Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Rope Ferry Road
Waterford, CT 06385

Dominion

JUL 15 2010

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Serial No. 10-425
MPS Lic/LES R0
Docket No. 50-423
License No. NPF-49

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3
LICENSEE EVENT REPORT 2010-002-00
MPS3 AUTOMATIC REACTOR TRIP ON LO LO C STEAM GENERATOR WATER LEVEL

This letter forwards Licensee Event Report (LER) 2010-002-00 documenting an event that occurred at Millstone Power Station Unit 3, on May 17, 2010. This LER provides the follow-up report to an event was reported in accordance with 10 CFR 50.73 (a)(2)(iv) via event notification 45931 pursuant to 10 CFR 50.72 (b)(2)(iv)(B).

If you have any questions or require additional information, please contact Mr. William D. Bartron at (860) 444-4301.

Sincerely,

A. J. Jordan
Site Vice President – Millstone

Attachments: 1

Commitments made in this letter: None

IE22
NRR

cc: U.S. Nuclear Regulatory Commission
Region I
475 Allendale Road
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NRC Senior Resident Inspector
Millstone Power Station

ATTACHMENT

LICENSEE EVENT REPORT 2010-002-00

MILLSTONE POWER STATION
DOMINION NUCLEAR CONNECTICUT, INC.

NRC FORM 366 (9-2007) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104 EXPIRES: 08/31/2010
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory collection request: 80 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Millstone Power Station - Unit 3	05000423	1 OF 3

4. TITLE
Automatic Reactor Trip on Lo Lo Steam Generator Level

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
05	17	2010	2010	002	00	07	15	2010	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply)			
1	☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
	☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
	☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
	☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
10. POWER LEVEL	☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	X 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
017	☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
	☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
	☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
	☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME	TELEPHONE NUMBER (Include Area Code)
William D. Bartron, Nuclear Station Licensing	860-444-4301

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX

14. SUPPLEMENTAL REPORT EXPECTED		15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES (If yes, complete 15. EXPECTED SUBMISSION DATE)	☒ NO				

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

On May 17, 2010 an automatic reactor trip occurred on Millstone Power Station Unit 3 (MPS3) while the unit was in Mode 1, at 17 % power, due to a low steam generator (S/G) water level condition in 'C' S/G caused by the inability of the feedwater regulating bypass valve (FRBV) to properly control S/G levels in automatic or manual control. All control rods fully inserted into the reactor. The auxiliary feedwater system started as designed and maintained S/G level. Safety systems functioned as expected based upon the signals received. There were no radiological challenges as a result of the event.

The cause of this event was determined to be the inability of the FRBV control system to control S/G levels at low power operations. The control loop was dynamically tuned and the level controller adjusted to improve system control and performance. A design change is being evaluated to improve the performance of the SG level control system during low power operations.

This event is being reported in accordance with 10 CFR 50.73(a)(2)(iv)(A) as an event that resulted in manual or automatic actuation of systems listed in 10 CFR 50.73(a)(2)(iv)(B).

NARRATIVE

1. Event Description

On May 17, 2010 an automatic reactor trip occurred on Millstone Power Station Unit 3 (MPS3) while the unit was in Mode 1, at 17 % power, due to a low steam generator (S/G) [SB] [SG] water level condition in 'C' S/G caused by slow response of the feedwater (FW) regulating bypass valve [V] in automatic and manual control.
The event began with the FW regulating bypass valves in automatic control. Water level oscillations were observed in the 'B' and 'C' S/Gs between 40% and 60% indicated level. Just prior to the event, the FW regulating bypass valves were taken from automatic to manual control to permit scheduled adjustments to power range nuclear instruments. Operators adjusted the 'C' bypass valve controller to attempt to smooth out level oscillations in the 'C' S/G. The oscillations were not able to be dampened out, and shrink caused by the introduction of cooler water caused the 'C' S/G water level to decrease to the lo lo S/G water level reactor trip setpoint (18.1%), and an automatic reactor trip occurred.

All control rods fully inserted into the reactor [AB] [RCT]. The auxiliary feedwater (AFW) system [BA] started as designed and maintained S/G level. All other post trip actions were completed as specified. Safety systems functioned as expected based upon the signals received. There were no radiological challenges as a result of the event.

This event is being reported in accordance with 10 CFR 50.73(a)(2)(iv)(A) as an event that resulted in manual or automatic actuation of the reactor protective system and the auxiliary feedwater system.

2. Cause

Inadequate design of the system that controls MPS3 S/G levels at low power operations was determined to be the cause of this event. The design allows for excessive oscillation in S/G water levels while in automatic control and equipment challenges while in manual control.

3. Assessment of Safety Consequences

There were no safety consequences associated with this event. All control rods inserted following the reactor trip on S/G level. The operating crew responded to the reactor trip by entering Emergency Operating Procedure (EOP) 35 E-0, "Reactor Trip or Safety Injection". The turbine generator [TA] [TB] had not been synchronized to the grid prior to the trip. The turbine bypass valves were controlling steam pressure.

Following the reactor trip, the turbine bypass valves continued to control main steam [SB] pressure and the main steam safety valves [RV] were not challenged. The AFW system started automatically on the trip as expected, and restored the S/G levels to their normal operating band, maintaining reactor coolant system (RCS) heat removal. With the reactor at 17% power following the completion of a refueling outage, RCS decay heat levels were low. As a result, a cooldown of the RCS to approximately 543 degrees F occurred. This is 14 degrees below the typical no load temperature of 557 degrees F. This slight cooldown was terminated by operator action to manually trip the turbine driven main FW pump in accordance with EOP ES-0.1, "Reactor Trip Response". There was no safety injection system actuation.

Operator actions and plant mitigating equipment responses were as expected with no safety system failures. There were no challenges to the fuel, RCS or containment fission product barriers.

1. FACILITY NAME	2. DOCKET	6. LER NUMBER			3. PAGE
		YEAR	SEQUENTIAL NUMBER	REV NO.	
Millstone Power Station - Unit 3	05000423	2010	- 002 -	00	3 OF 3

NARRATIVE

4. Corrective Action

The control loop was dynamically tuned prior to restart. Additional short-term corrective actions included adjusting the level controller to improve valve positioner performance and conducting additional just-in-time training for the operators.

Operations Procedure OP-3203 has been revised to perform calorimetric calibration of nuclear instruments at a power level during plant startup when the FW system is in a more stable condition.

A design change is being evaluated to improve the performance of the S/G level control system at low power.

5. Previous Occurrences

There were no previous similar events found.

Energy Industry Identification System (EIIS) codes are identified in the text as [XX].